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High Temperature Stress Analysis on 61-pin Test Assembly for Reactor Core Sub-channel Flow Test

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1. Introduction

Securing the structural integrity of a fuel assembly during reactor operation is of utmost importance in order to prevent reactor severe accident like the Fukushima nuclear power plant through a flow characteristics tests with test assembly scaled down from a prototype reactor of a sodium-cooled fast reactor (SFR)(1) and an integrity evaluation. Thus high temperature structural design and integrity evaluation are important to prevent potential damage of fuel subassembly and reactor core.

In this study, a high temperature heat transfer and stress analysis of a 61-pin test fuel assembly scaled down from the full scale 217-pin sub-assembly was conducted. The reactor core subchannel flow characteristic test will be conducted to evaluate uncertainties in computer codes used for reactor core thermal hydraulic design. Stress analysis for a 61-pin fuel assembly scaled down from Prototype Generation IV Sodium-cooled Fast Reactor (PGSFR) was conducted and structural integrity in terms of load controlled stress limits was conducted.

2. Test fuel assembly for Reactor core sub-channel flow test

The test loop for reactor core sub-channel flow test is shown in Fig. 1, and the 61-pin test fuel assembly is to be installed at the central test section part. The test loop is named as 'FIFFA' (Flow Identification test loop for Fast reactor Fuel Assembly). Fig. 2 shows the configuration of subassembly and test facility set-up for iso-kinetic flow rate sampling for 37-pin test assembly installed in FIFFA.

Design parameters of test assembly (61-pin) and subassembly of PGSFR of 150MWe capacity (217-pin) are shown in Table 1. The fuel assemblies of a prototype reactor and test section are hexagonal type and are composed of a triangular lattice type structure. The test section part is located at lower part in right hand side image in Fig. 2.

In scaling of test assembly from prototype SFR, it is important to preserve the geometric similarity conditions for pitch to diameter (P/D) and lead length to diameter (H/D). As shown in Table 1, those two parameters were preserved in design of test subassembly and the material of test section is stainless steel 316L.

In this study, the whole body of test assembly including fuel fin, wire wrap and outer casings was taken into account in the finite element modeling. Analyses on heat transfer and stress were conducted assuming the test assembly is installed in reactor core of PGSFR so that it is to be exposed to maximum sodium temperature of 554.4°C.

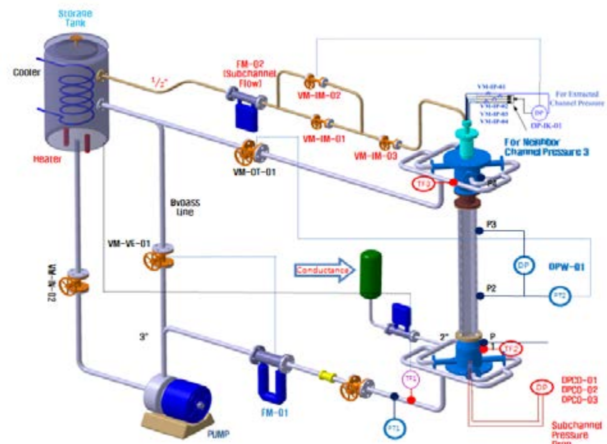

Fig. 1 P&ID of sub-channel flow characteristics test facility

Fig. 2 Images of sub-assembly and test facility set-up

Table 1 Design parameters of fuel sub-assembly

	PGSFR	Test Assembly
Number of fuel pin	217-pin	61-pin
Inner to inner wall distance(mm)	126	73.3
Pin OD (mm)	7.4	8.0
Wire Wrap OD (mm)	0.95	1.03
Wire Wrap Pitch (mm)	204.9	221.5
Fuel Element Length	2150+ α	1500
P/D Ratio	1.14	
H/D Ratio	27.69	

3. 3D Finite element analysis

A full 3-D finite element analysis was conducted for the 61-pin test assembly. The FE model is is composed of 10,642 3D linear solid and 926,861 nodes. As the boundary condition for the test fuel assembly, the bottom surface of the 61-pin test assembly was completely fixed while thermal expansion in the radial direction was allowed. The maximum temperature of sodium in reactor core sub-channel was shown to be maximum 554.4°C. The 61-pin test assembly was assumed to be exposed to hot sodium of maximum 554.4°C. The design transients of the test assembly are composed of 5 load steps ; steady state 554.4°C, cool-down to 254.4° with cool-down rate of 100°C/hr, steady state at 254.4°C, heat-up to 554.4°C and finally reaching 554.4°C as shown in Fig.3.

The heat transfer coefficient was assumed to be uniformly 50,000 m^2K . The inlet temperature at the bottom of the test assembly was set to be 390°C and outlet of rod bundle to be 554.4°C. The sodium temperatures in between the inlet and outlet were interpolated linearly and heat convection coefficients were taken into account for the heat transfer analysis.

Heat transfer analysis was conducted and temperature distributions for the 61-pin fuel assembly were obtained as in Fig. 4 at the end of the heat-up.

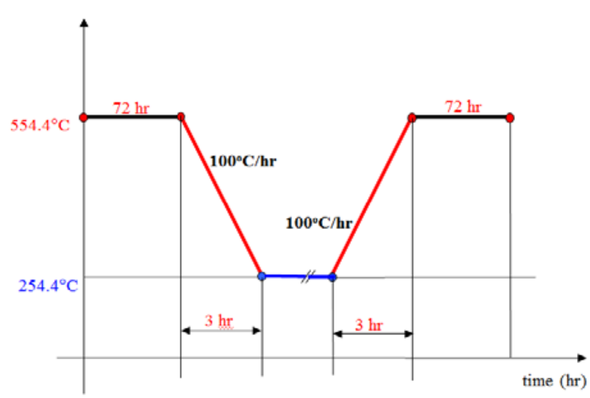

Fig.3 Thermal loading conditions in test fuel assembly

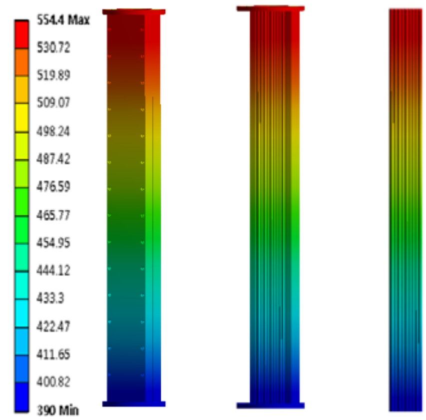

Fig. 4 Temperature distribution at the end heat-up

Fig. 4 shows temperature profiles of gradual change in temperature distributions along the axial direction in a test assembly. Evaluations on load-controlled stress limits were conducted according to the elevated design codes[2,3]

4. High temperature load-controlled stress limits

High temperature integrity evaluations were performed for the test assembly based of 3-D element analysis.

Stress analysis results under primary load showed that maximum stress intensity (S.I) of 4.05MPa occurred at mid part of the test assembly as shown in Fig. 5. Comparing the value of 4.05MPa with the code allowable[3] of 127MPa, it was shown that the stress intensity due to due to primary load is very small.

Stress analysis results under primary and secondary loads showed that the maximum stress intensity of 84.08MPa occurred at upper flange tangent to outer casing as shown in Fig. 6. Comparing the value of 84.08MPa with the code allowable of 268.8MPa, it was shown that the stress intensity due to due to primary plus secondary stress range was well within the allowable. Tresca S.I is comparison target with the design code limit values but it is to be noted that Tresca S.I is slightly different from von Mises S.I.

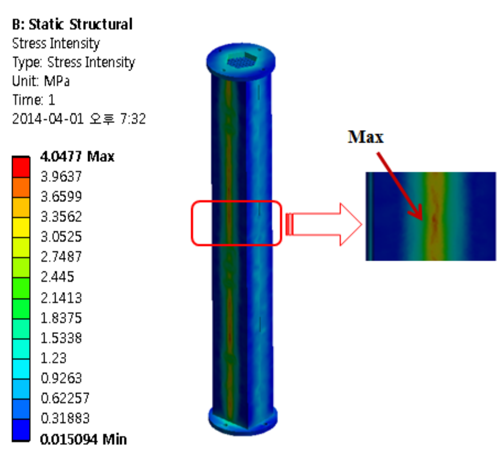

Fig. 6 Mises stress intensities under primary loads

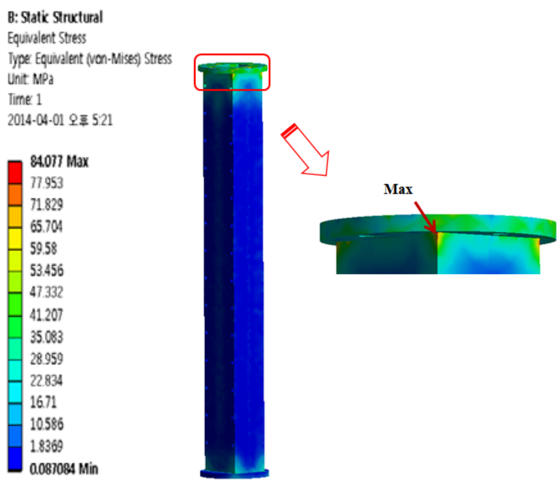

Fig. 7 Mises stress intensity range under secondary loads

The integrity evaluations based on strain limits and creep-fatigue damage should be conducted according to the elevated design codes for the test fuel assembly. The calculations are currently underway according to the design codes of ASME-NH(2) and RCC-MRx(3).

5. Conclusions

In this study, The evaluations on load-controlled stress limits for a 61-pin test fuel assembly to be used for reactor core subchannel flow distribution tests were conducted assuming that the test assembly is installed in a Prototype Generation IV Sodium-cooled fast reactor core. The 61-pin test assembly has the geometric similarity on P/D and H/D with PGSFR and material of fuel assembly is austenitic stainless steel 316L.

The stress analysis results showed that 4.05MPa under primary load occurred at mid part of the test assembly and it was shown that the value of 4.05Mpa was far smaller than the code allowable(3) of 127MPa. , it was shown that the stress intensity due to due to primary load is very small. The stress analysis results under primary and secondary loads showed that maximum stress intensity of 84.08MPa occurred at upper flange tangent to outer casing and the value was well within the code allowable of 268.8MPa. Integrity evaluations based on strain limits and creep-fatigue damage are underway according to the elevated design codes.

Acknowledgements

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[3] RCC-MRx, Section III Subsection B, Class 1 N1RX, 2012, Reactor Components its Auxiliary Systems and Supports, 2012 Edition, AFCEN.